Westinghouse Non-Proprietary Class 3

WCAP-16306-NP
Revision 0

August 2004

Evaluation of Pressurized Thermal Shock for V. C. Summer

Westinghouse

WCAP-16306-NP Revision 0

Evaluation of Pressurized Thermal Shock for V. C. Summer

C. M. Burton

AUGUST 2004

Approved: John Ghergurovich
John Ghergurovich, Manager
Reactor Component Design & Analysis

Westinghouse Electric Company LLC
Energy Systems
P. O. Box 355
Pittsburgh, PA 15230-0355

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

PREFACE

This report has been technically reviewed and verified by:

Reviewer: T. J. Laubham

EXECUTIVE SUMMARY

The purpose of this report is to determine the RT_{PTS} values for the V. C. Summer reactor vessel beltline based upon the results of the Surveillance Capsule Z evaluation. The conclusion of this report is that all the beltline materials in the V. C. Summer reactor vessel have RT_{PTS} values below the screening criteria of 270°F for plates, forgings or longitudinal welds and 300°F for circumferential welds at EOL (End of Life) and EOLE (End of Life Extension), 32 and 56 EFPY respectively.

1. INTRODUCTION

A Pressurized Thermal Shock (PTS) Event is an event or transient in pressurized water reactors (PWRs) causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel. A PTS concern arises if one of these transients acts on the beltline region of a reactor vessel where a reduced fracture resistance exists because of neutron irradiation. Such an event may produce a flaw or cause the propagation of a flaw postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

The purpose of this report is to determine the RT_{PTS} values for the V. C. Summer reactor vessel incorporating the results of the surveillance Capsule Z evaluation. Section 2 discusses the PTS Rule and its requirements. Section 3 provides the methodology for calculating RT_{PTS}. Section 4 provides the reactor vessel beltline region material properties for the V. C. Summer reactor vessel. The neutron fluence values used in this analysis are presented in Section 5.0 and were obtained from WCAP-16298-NP[8.1]. The results of the RT_{PTS} calculations are presented in Section 6. The conclusion and references for the PTS evaluation follow in Sections 7 and 8, respectively.

2. PRESSURIZED THERMAL SHOCK RULE

The Nuclear Regulatory Commission (NRC) amended its regulations for light-water-cooled nuclear power plants to clarify several items related to the fracture toughness requirements for reactor pressure vessels, including pressurized thermal shock requirements. The revised PTS Rule[8.2], 10 CFR Part 50.61, was published in the Federal Register on December 19, 1995, with an effective date of January 18, 1996.

This amendment to the PTS Rule makes the following changes:

- The rule incorporates in total, and therefore makes binding by rule, the method for determining the reference temperature, RT_{NDT}, including treatment of the unirradiated RT_{NDT} value, the margin term, and the explicit definition of "credible" surveillance data, which is currently described in Regulatory Guide 1.99, Revision 2[8.3].

- The rule is restructured to improve clarity, with the requirements section giving only the requirements for the value for the reference temperature for end of life (EOL) fluence, RT_{PTS}.

- Thermal annealing is identified as a method for mitigating the effects of neutron irradiation, thereby reducing RT_{PTS}.

The PTS Rule requirements consist of the following:

- For each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall have projected values of RT_{PTS}, accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material.

- The assessment of RT_{PTS} must use the calculation procedures given in the PTS Rule, and must specify the bases for the projected value of RT_{PTS} for each beltline material. The report must specify the copper and nickel contents and the fluence values used in the calculation for each beltline material.

- This assessment must be updated whenever there is significant change in projected values of RT_{PTS} or upon the request for a change in the expiration date for operation of the facility. Changes to RT_{PTS} values are significant if either the previous value or the current value, or both values, exceed the screening criterion prior to the expiration of the operating license, including any renewal term, if applicable for the plant.

- The RT_{PTS} screening criterion values for the beltline region are:

  - 270°F for plates, forgings and axial weld materials
  - 300°F for circumferential weld materials

- All available surveillance data must be considered in the evaluation. All credible plant specific surveillance data must also be used in the evaluation.

3. METHOD FOR CALCULATION OF RT_{PTS}

RT_{PTS} must be calculated for each vessel beltline material using a fluence value, f, which is the EOL fluence for the material. Equation 1 must be used to calculate values of RT_{NDT} for each weld and plate or forging in the reactor vessel beltline.

$$RT_{NDT} = RT_{NDT(U)} + M + \Delta RT_{NDT} \qquad (1)$$

Where,

$RT_{NDT(U)}$ = Reference Temperature for a reactor vessel material in the pre-service or unirradiated condition

M = Margin to be added to account for uncertainties in the values of $RT_{NDT(U)}$, copper and nickel contents, fluence and calculational procedures. M is evaluated from Equation 2

$$M = 2 * \sqrt{\sigma_U^2 + \sigma_\Delta^2} \qquad (2)$$

σ_U is the standard deviation for $RT_{NDT(U)}$

σ_U = 0°F when $RT_{NDT(U)}$ is a measured value

σ_U = 17°F when $RT_{NDT(U)}$ is a generic value

σ_Δ is the standard deviation for RT_{NDT}

For plates and forgings:

σ_Δ = 17°F when surveillance capsule data is not used

σ_Δ = 8.5°F when surveillance capsule data is used

For welds:

σ_Δ = 28°F when surveillance capsule data is not used

σ_Δ = 14°F when surveillance capsule data is used

σ_Δ not to exceed one half of ΔRT_{NDT}

ΔRT_{NDT} is the mean value of the transition temperature shift, or change in ΔRT_{NDT}, due to irradiation, and must be calculated using Equation 3.

$$\Delta RT_{NDT} = (CF) * f^{(0.28 - 0.10 \log f)} \qquad (3)$$

"CF" (°F) is the chemistry factor, which is a function of copper and nickel content. CF is determined from Tables 1 and 2 of the PTS Rule (10 CFR 50.61[8.2]). Surveillance data deemed credible must be used to determine a material-specific value of CF. A material-specific value of CF is determined in Equation 5.

"f" is the calculated neutron fluence, in units of 10^{19} n/cm^2 (E > 1.0 MeV), at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence. The EOL fluence is used in calculating RT_{PTS}.

Equation 4 must be used for determining RT_{PTS} using Equation 3 with EOL fluence values for determining ΔRT_{PTS}

$$RT_{PTS} = RT_{NDT(U)} + M + \Delta RT_{PTS} \qquad (4)$$

To verify that RT_{NDT} for each vessel beltline material is a bounding value for the specific reactor vessel, licensees shall consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to the reactor vessel operating temperature and any related surveillance program results. Results from the plant-specific surveillance program must be integrated into the RT_{NDT} estimate if the plant-specific surveillance data has been deemed credible.

A material-specific value of CF is determined from Equation 5.

$$CF = \frac{\sum[A_i * f_i^{(0.28-0.20\log f_i)}]}{\sum[f_i^{(0.56-0.20\log f_i)}]} \qquad (5)$$

In Equation 5, "A_i" is the measured value of ΔRT_{NDT} and "f_i" is the fluence for each surveillance data point. If there is clear evidence that the copper and nickel content of the surveillance weld differs from the vessel weld, i.e., differs from the average for the weld wire heat number associated with the vessel weld and the surveillance weld, the measured values of RT_{NDT} must be adjusted for differences in copper and nickel content by multiplying them by the ratio of the chemistry factor for the vessel material to that for the surveillance weld.

4. VERIFICATION OF PLANT SPECIFIC MATERIAL PROPERTIES

Before performing the pressurized thermal shock evaluation, a review of the latest plant-specific material properties for the V. C. Summer vessel was performed. The beltline region of a reactor vessel, per the PTS Rule, is defined as "the region of the reactor vessel (shell material including welds, heat-affected zones and plates and forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage". Figure 1 identifies and indicates the location of all beltline region materials for the V. C. Summer reactor vessel.

The best estimate copper and nickel contents of the beltline materials were obtained from V. C. Summer's response to NRC Generic Letter 92-01 Revision 1, Supplement 1[8.4], WCAP-12867[8.5] and CE Report NPSD-1039, Rev. 2[8.6]. The best estimate copper and nickel content is documented in Table 1 herein. The average values were calculated using all of the available and appropriate material chemistry information. Initial RT_{NDT} values for V. C. Summer reactor vessel beltline material properties are also shown in Table 1.

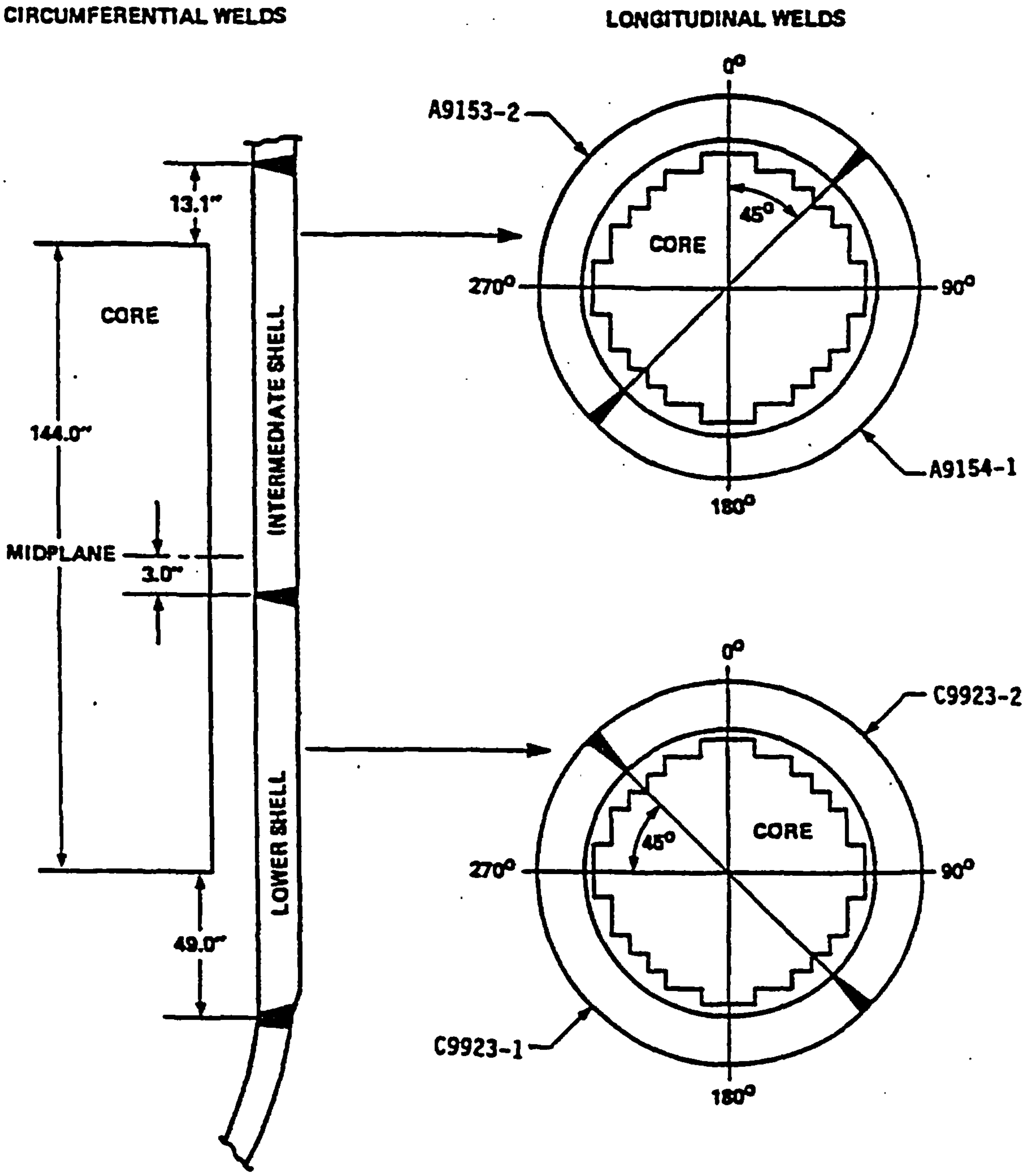

FIGURE 1 Identification and Location of Beltline Region Materials for the V. C. Summer Reactor Vessel

TABLE 1
V. C. Summer Reactor Vessel Beltline Unirradiated Material Properties

Material Description	Cu (%)	Ni(%)	Initial RT_{NDT}[(a)]
Closure Head Flange 5297-V1[(b)]	- -	- -	10°F[(b)]
Vessel Flange 5301-V-1	- -	- -	0°F[(b)]
Intermediate Shell Plate A9154-1	0.10	0.51	30°F
Intermediate Shell Plate A9153-2	0.09	0.45	-20°F
Lower Shell Plate C9923-1	0.08	0.41	10°F
Lower Shell Plate C9923-2	0.08	0.41	10°F
Intermediate Shell Longitudinal Weld Seams BC & BD	0.05	0.91	-44°F
Intermediate Shell Longitudinal Weld Seams BA & BB	0.05	0.91	-44°F
Intermediate to Lower Shell Plate Circumferential Weld Seam AB	0.05	0.91	-44°F
Surveillance Program Weld Metal	0.04	0.95	- - -

(a) The initial RT_{NDT} values for the plates and welds are based on measured data per WCAP-12867[8.5]

(b) In the past the closure head flange was reported as Heat A9231 with an IRT_{NDT} of -20°F. Based on a review of Westinghouse files, the correct data is Heat # 5297-V1 with an IRT_{NDT} of 10°F. Also, the vessel flange reported an IRT_{NDT} of 10°F., however, based on a review Westinghouse files, the correct IRT_{NDT} is 0°F.

5. NEUTRON FLUENCE VALUES

The calculated fast neutron fluence (E > 1.0 MeV) values at the inner surface of the V. C. Summer reactor vessel for 32 and 56 EFPY are shown in Table 2. These values were projected using ENDF/B-VI cross sections and are based on the results of the Capsule X radiation analysis and comply with Reg. Guide 1.190[8.8]. Note that the fluence projections were obtained from WCAP-16298-NP[8.1] (Capsule Z analysis report).

TABLE 2
Calculated Fluence (E > 1.0 MeV) on the Pressure Vessel Clad/Base Interface for V. C. Summer at 32 (EOL) and 56 (EOLE) EFPY

Material	Location(a)	32 EFPY Fluence	56 EFPY Fluence
Intermediate Shell Plate A9154-1	0°	3.92×10^{19}	6.80×10^{19}
Intermediate Shell Plate A9153-2	0°	3.92×10^{19}	6.80×10^{19}
Lower Shell Plate C9923-1	0°	3.92×10^{19}	6.80×10^{19}
Lower Shell Plate C9923-2	0°	3.92×10^{19}	6.80×10^{19}
Circumferential Weld Seam AB	0°	3.92×10^{19}	6.80×10^{19}
Longitudinal Weld Seams BC, BD, BA & BB	45°	1.36×10^{19}	2.35×10^{19}

(a) These locations are shown graphically in Figure 1

Calculated Integrated Neutron Exposure of the Surveillance Capsules @ V. C. Summer are shown in Table 3.

TABLE 3
Calculated Integrated Neutron Exposure of the Surveillance Capsules @ V. C. Summer

CAPSULE	FLUENCE
U	6.77×10^{18} n/cm^2, (E > 1.0 MeV)
V	1.56×10^{19} n/cm^2, (E > 1.0 MeV)
X	2.53×10^{19} n/cm^2, (E > 1.0 MeV)
W	4.63×10^{19} n/cm^2, (E > 1.0 MeV)
Z	6.54×10^{19} n/cm^2, (E > 1.0 MeV)

6. DETERMINATION OF RT_{PTS} VALUES FOR ALL BELTLINE REGION MATERIALS

Using the prescribed PTS Rule methodology, RT_{PTS} values were generated for all beltline region materials of the V. C. Summer reactor vessel for fluence values at the EOL (32 EFPY) and EOLE (56 EFPY).

Each plant shall assess the RT_{PTS} values based on plant-specific surveillance capsule data. For V. C. Summer, the related surveillance program results have been included in this PTS evaluation. Tables 6 and 7 contain the RT_{PTS} calculations for all beltline region materials at EOL (32 EFPY) and EOLE (56 EFPY).

The chemistry factors were calculated using Regulatory Guide 1.99 Revision 2, Positions 1.1 and 2.1. Position 1.1 uses the Tables from the Reg. Guide along with the best estimate copper and nickel weight percents. Position 2.1 uses the surveillance capsule data from all capsules withdrawn to date. The fluence values used to determine the CFs in Table 4 are the calculated fluence values at the surveillance capsule locations. Hence, the calculated fluence values were used for all cases. The measured ΔRT_{NDT} values for the weld data were adjusted for differences between chemistry of the surveillance weld and the vessel weld using the ratio procedure given in Position 2.1 of Regulatory Guide 1.99, Revision 2.

Credibility Evaluation

V. C. Summer surveillance program contains surveillance material from the Intermediate Shell Plate A9154-1 and from weld material fabricated from weld wire heat number 4P4784, which is the same heat as all the beltline weld seams. In order to apply the surveillance data to such evaluations as PTS, you first have to evaluate whether or not the data is credible. The procedures for evaluating credibility are prescribed in 10CFR50.61, along with guidance provided by the NRC at an industry meeting on February 12th & 13th, 1998. The credibility evaluation for the V. C. Summer data has already been performed under the Capsule Z report, WCAP-16298-NP[8.1], and the results were as follows:

- Surveillance plate material from Intermediate Shell Plate A9154-1 was determined to be not credible
- Surveillance weld metal (Heat 4P4784) was determined to be credible.

TABLE 4

Calculation of Chemistry Factors using V. C. Summer Surveillance Capsule Data

Material	Capsule	Capsule $f^{(a)}$	$FF^{(b)}$	$\Delta RT_{NDT}{}^{(c)}$	$FF^*\Delta RT_{NDT}$	FF^2
Intermediate Shell Plate A9154-1 (Longitudinal)	U	0.677	0.891	36.1	32.2	0.793
	V	1.56	1.123	53.2	59.7	1.261
	X	2.53	1.249	38.3	47.8	1.560
	W	4.63	1.387	66.2	91.8	1.924
	Z	6.54	1.452	98.9	143.6	2.108
Intermediate Shell Plate A9154-1 (Transverse)	U	0.677	0.891	14.5	12.9	0.793
	V	1.56	1.123	32.1	36.0	1.261
	X	2.53	1.249	26.7	33.3	1.560
	W	4.63	1.387	57.8	80.2	1.924
	Z	6.54	1.452	87.0	126.3	2.108
				SUM:	663.8	15.292
	$CF_{A9154\text{-}1} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (663.8) \div (15.292) =$ **43.4°F**					
Surveillance Weld Material$^{(d)}$	U	0.677	0.891	28.6 (22.7)$^{(d)}$	25.4	0.793
	V	1.56	1.123	59.2 (47.0)$^{(d)}$	66.5	1.261
	X	2.53	1.249	28.6 (22.7)$^{(d)}$	35.7	1.560
	W	4.63	1.387	54.8 (43.5)$^{(d)}$	76.0	1.924
	Z	6.54	1.452	82.2 (65.2)$^{(d)}$	119.3	2.108
				SUM:	323.0	7.646
	$CF_{Surv.\,Weld} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (323.0) \div (7.646) =$ **42.2°F**					

(a) f = fluence. See Table 3 [x 10^{19} n/cm^2, E > 1.0 MeV]

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values given in the Capsule Z analysis report, WCAP-16298-NP[8.1], [°F]

(d) The Surveillance Weld ΔRT_{NDT} values have been adjusted by a ratio of 0.671 ($CF_{VW} \div CF_{SW}$ = 32.6 ÷ 48.6), the pre-adjusted values are in parenthesis

TABLE 5

Summary of the V. C. Summer Reactor Vessel Beltline Material Chemistry Factors

Material	Reg. Guide 1.99, Rev. 2 Position 1.1 CF's	Reg. Guide 1.99, Rev. 2 Position 2.1 CF's
Intermediate Shell Plate A9154-1	65.0°F	43.4°F[(a)]
Intermediate Shell Plate A9153-2	58.0°F	---
Lower Shell Plate C9923-1	51.0°F	---
Lower Shell Plate C9923-2	51.0°F	---
Intermediate Shell Longitudinal Weld Seams BC & BD	68.0°F	42.2°F[(a)]
Lower Shell Longitudinal Weld Seams BA & BB	68.0°F	42.2°F[(a)]
Intermediate to Lower Shell Plate Circumferential Weld Seam AB	68.0°F	42.2°F[(a)]

(a) See Capsule Z Report, WCAP-16298-NP[8.1], for the credibility evaluation of the V.C. Summer Unit 1 surveillance data. The Intermediate Shell Plate A9154-1 was deemed "non-credible" while the weld was deemed "credible".

TABLE 6

RT_{PTS} Calculation for V. C. Summer Beltline Region Materials at EOL (32 EFPY)

Material	Fluence, f (n/cm^2, E>1.0 MeV)(a)	FF(b)	CF (°F) (c)	ΔRT_{PTS} (°F) (d)	Margin (°F) (e)	$RT_{NDT(U)}$ (°F) (f)	RT_{PTS} (°F) (g)
Intermediate Shell Plate A9154-1	3.92×10^{19}	1.352	65.0	87.9	34	30	152
Intermediate Shell Plate A9154-1 Using S/C Data	3.92×10^{19}	1.352	43.4	58.7	34(h)	30	123
Intermediate Shell Plate A9153-2	3.92×10^{19}	1.352	58.0	78.4	34	-20	92
Lower Shell Plate C9923-1	3.92×10^{19}	1.352	51.0	69.0	34	10	113
Lower Shell Plate C9923-2	3.92×10^{19}	1.352	51.0	69.0	34	10	113
Intermediate to Lower Shell Circumferential Weld Seam AB	3.92×10^{19}	1.352	68.0	91.9	56	-44	104
Intermediate to Lower Shell Circumferential Weld Seam AB Using S/C Data	3.92×10^{19}	1.352	42.2	57.1	28	-44	41
Intermediate and Lower Shell Longitudinal Weld Seams BC, BD, BA & BB (45° Azimuth)	1.36×10^{19}	1.085	68.0	73.8	56	-44	86
Intermediate and Lower Shell Longitudinal Weld Seams BC, BD, BA & BB (45° Azimuth) Using S/C Data	1.36×10^{19}	1.085	42.2	45.8	28	-44	30

(a) The fluence, f, was taken from the peak azimuthal location, see Table 2

(b) $FF = f^{(.28 - 0.1 * \log f)}$, where f is the clad/base metal interface fluence

(c) Chemistry Factor is taken from Table 5

(d) $\Delta RT_{PTS} = CF * FF$

(e) Margin = $2*(\sigma_u^2 + \sigma_\Delta^2)^{1/2}$, see Section 3

(f) Initial RT_{NDT} values are measured values, see Table 1

(g) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F) (This value was rounded per ASTM E29, using the "Rounding Method")

(h) Data "not credible", per [8.1]

TABLE 7

RT_{PTS} Calculation for V. C. Summer Beltline Region Materials at EOLE (56 EFPY)

Material	Fluence, f (n/cm^2, E>1.0 MeV)[a]	FF[b]	CF (°F) [c]	ΔRT_{PTS} (°F) [d]	Margin (°F) [e]	$RT_{NDT(U)}$ (°F) [f]	RT_{PTS} (°F) [g]
Intermediate Shell Plate A9154-1	6.80×10^{19}	1.458	65.0	94.8	34	30	159
Intermediate Shell Plate A9154-1 Using S/C Data	6.80×10^{19}	1.458	43.4	63.3	34[h]	30	127
Intermediate Shell Plate A9153-2	6.80×10^{19}	1.458	58.0	84.6	34	-20	99
Lower Shell Plate C9923-1	6.80×10^{19}	1.458	51.0	74.4	34	10	118
Lower Shell Plate C9923-2	6.80×10^{19}	1.458	51.0	74.4	34	10	118
Intermediate to Lower Shell Circumferential Weld Seam AB	6.80×10^{19}	1.458	68.0	99.1	56	-44	111
Intermediate to Lower Shell Circumferential Weld Seam AB Using S/C Data	6.80×10^{19}	1.458	42.2	61.5	28	-44	46
Intermediate and Lower Shell Longitudinal Weld Seams BC, BD, BA & BB (45° Azimuth)	2.35×10^{19}	1.231	68.0	83.7	56	-44	96
Intermediate and Lower Shell Longitudinal Weld Seams BC, BD, BA & BB (45° Azimuth) Using S/C Data	2.35×10^{19}	1.231	42.2	51.9	28	-44	36

(a) The fluence, f, was taken from the peak azimuthal location, see Table 2

(b) $FF = f^{(.28 - 0.1*\log f)}$, where f is the clad/base metal interface fluence

(c) Chemistry Factor is taken from Table 5

(d) $\Delta RT_{PTS} = CF * FF$

(e) Margin = $2*(\sigma_u^2 + \sigma_\Delta^2)^{1/2}$, see Section 3

(f) Initial RT_{NDT} values are measured values, see Table 1

(g) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F) (This value was rounded per ASTM E29, using the "Rounding Method")

(h) Data "not credible", per [8.1]

7. CONCLUSION

As shown in Tables 6 and 7, all of the beltline region materials in the V. C. Summer reactor vessel have EOL (32 EFPY) RT_{PTS} and EOLE (56 EFPY) RT_{PTS} values below the screening criteria values of 270°F for plates, forgings and longitudinal welds and 300°F for circumferential welds.

8. REFERENCES

8.1 WCAP-16298-NP, "Analysis of Capsule Z from the South Carolina Electric & Gas Company V. C. Summer Reactor Vessel Radiation Surveillance Program", C. M. Burton, et.al., August 2004

8.2 10 CFR Part 50.61, "Fracture Toughness Requirements For Protection Against Presuurized Thermal Shock Events", Federal Register, Volume 60, No. 243, dated December 19, 1995

8.3 Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", U.S. Nuclear Regulatory Commission, May, 1988

8.4 South Carolina Electric and Gas Letter, G. J. Taylor to U. S. NRC, Dated 11/7/95, Subject: "Virgil C. Summer Nuclear Station, Docket No. 50/395, Operating License No. NPF-12, Response to G. L. 92-01, Revision 1, Supplement 1"

8.5 WCAP-12867, "Analysis of Capsule X from the South Carolina Electric & Gas Company Virgil C. Summer Unit 1 Reactor Vessel Radiation Surveillance Program", J. M. Chicots, et. al., March 1991

8.6 CE NPSD-1039, Rev. 2, "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds, Appendix A, CE Reactor Vessel Weld Properties Database, Volume 1", CEOG Task 902, June 1997

8.7 Westinghouse Calculation 92-016, Page 53 of 176 to 57 of 176, "WOG USE Program – Onset of Upper Shelf Energy Calculations", J. M. Chicots, 11/12/92

8.8 Regulatory Guide RG-1.190, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence*, U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, March 2001

APPENDIX A

PREDICTED EOL USE VALUES

Per Regulatory Guide 1.99, Revision 2, the Charpy upper-shelf energy is assumed to decrease as a function of fluence and copper content as indicated in Figure 2 (Figure A-1 of this report) of the guide when surveillance data is not used. Linear interpolation is permitted. In addition, if surveillance data is to be used, the decrease in upper-shelf energy may be obtained by plotting the reduced plant surveillance data on Figure 2 of the guide (Figure A-1 of this report) and fitting the data with a line drawn parallel to the existing lines as the upper bound of all the data. This line should be used in preference to the existing graph.

The EOL (32 EFPY) and EOLE (56 EFPY) USE can be predicted using the corresponding ¼T fluence projection, the copper content of the beltline materials and/or the results of the capsules tested to date using Figure 2 in Regulatory Guide 1.99, Revision 2 (Figure A-1 of this report). The peak vessel clad/base metal interface fluence value was used to determine the EOL (32 EFPY) and EOLE (56 EFPY) USE of all the beltline materials. In determining the projected USE decrease the measured USE decreases for all vessel beltline materials were below the lower limit of Figure 2. Therefore, the lower limit was used to find the projected end of life USE decrease.

The V. C. Summer reactor vessel beltline region minimum thickness is 7.75 inches.

The calculation of the ¼T vessel surface fluence values at 32 and 56 EFPY for the beltline materials is contained in Table A-1 of this report.

TABLE A-1

EOL (32 EFPY) & EOLE (56 EFPY) 1/4T Fluence Values for all the V. C. Summer Beltline Materials

Material	Fluence @ 32 EFPY[(a)]	1/4T F @ 32 EFPY[(b)]	Fluence @ 56 EFPY[(a)]	1/4T F @ 56 EFPY[(b)]
Intermediate Shell Plate A9154-1	3.92	2.46	6.80	4.27
Intermediate Shell Plate A9153-2	3.92	2.46	6.80	4.27
Lower Shell Plate C9923-1	3.92	2.46	6.80	4.27
Lower Shell Plate C9923-2	3.92	2.46	6.80	4.27
Circumferential Weld Seam AB	3.92	2.46	6.80	4.27
Longitudinal Weld Seams BC, BD, BA & BB (45° Azimuth)	1.36	0.85	2.35	1.48

Notes:

(a) Fluence at the clad/base metal interface [x 10^{19} n/cm^2, E > 1.0 MeV]; see Table 2

(b) 1/4T F = F $e^{(-0.24\ x)}$, where "x" is the depth into the vessel wall (X = ¼·reactor vessel beltline thickness = 0.25 · 7.75 inches = 1.9375 inches), [x10^{19} n/cm^2, E > 1.0MeV]

TABLE A-2

Predicted End-of-License EOL (32 EFPY) USE Calculations for all the Beltline Region Materials

Material	Weight % of Cu	1/4T EOL Fluence (10^{19} n/cm^2)	Unirradiated USE[a] (ft-lbs)	Projected USE Decrease (%)	Projected EOL USE (ft-lbs)
Intermediate Shell Plate A9154-1 Using S/C Data	0.10	2.46	81	14	70
Intermediate Shell Plate A9153-2	0.09	2.46	107	24	81
Lower Shell Plate C9923-1	0.08	2.46	106	24	81
Lower Shell Plate C9923-2	0.08	2.46	92	24	70
Intermediate to Lower Shell Circumferential Weld Seam AB Using S/C Data	0.05	2.46	84	10	76
Intermediate and Lower Shell Longitudinal Weld Seams BC, BD, BA & BB (45° Azimuth) Using S/C Data	0.05	0.85	84	8	77

Notes:

(a) These values were obtained from V. C. Summer and documented in [8.7]. Note that they all match what is listed in the NRC database RVID2.

(b) No surveillance capsule data was used because it was not limiting.

TABLE A-3

Predicted End-of-License Extension EOLE (56 EFPY) USE Calculations for all the Beltline Region Materials

Material	Weight % of Cu	1/4T EOLE Fluence (10^{19} n/cm^2)	Unirradiated USE[a] (ft-lbs)	Projected USE Decrease (%)	Projected EOL USE (ft-lbs)
Intermediate Shell Plate A9154-1 Using S/C Data	0.10	4.27	81	16	68
Intermediate Shell Plate A9153-2	0.09	4.27	107	27	78
Lower Shell Plate C9923-1	0.08	4.27	106	27	77
Lower Shell Plate C9923-2	0.08	4.27	92	27	67
Intermediate to Lower Shell Circumferential Weld Seam AB Using S/C Data	0.05	4.27	84	11	75
Intermediate and Lower Shell Longitudinal Weld Seams BC, BD, BA & BB (45° Azimuth) Using S/C Data	0.05	1.48	84	9	76

Notes:

(a) These values were obtained from V. C. Summer and documented in [8.7]. Note that they all match what is listed in the NRC database RVID2.

(b) No surveillance capsule data was used because it was not limiting.

Figure A-1
Regulatory Guide 1.99, Revision 2
Predicted Decrease in Upper Shelf Energy as a Function of Copper and Fluence